NON-PUBLIC?: N
ACCESSION #: 8812210144
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Palo Verde Unit 2 PAGE: 1 OF 8

DOCKET NUMBER: 05000529

TITLE: Reactor Trip Due to Low Steam Generator Level
EVENT DATE: 11/16/88 LER #: 88-014-00 REPORT DATE: 12/14/88

OPERATING MODE: 1 POWER LEVEL: 010

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Timothy D. Shriver
Compliance Manager TELEPHONE: 602-393-2521

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: AB COMPONENT: ISV MANUFACTURER: B350
REPORTABLE TO NPRDS: NO

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At approximately 0237 MST on November 16, 1988 Palo Verde Unit 2 was in Mode 1 (POWER OPERATION) at approximately 10 percent power when a reactor trip occurred. Unit 2 was being shut down to identify and repair a reactor coolant system (RCS) leak, which was within Technical Specification limits for continued operation, when the trip occurred as a result of low steam generator water level. The reactor trip was uncomplicated and stable conditions were achieved at approximately 0247 MST terminating the event. There were no Engineered Safety Feature actuations and none were necessary.

The cause of the low steam generator water level was inadequate feedwater flow due to main feedwater pump speed being too slow for the existing plant conditions. The cause of the RCS leak was excess packing leakage as a result of a broken packing gland follower bolt on an instrument isolation valve.

As corrective action to prevent recurrence, procedural changes have been implemented, an evaluation of the feedwater pump control system is being performed, and programs for implementation of plant modifications will be

evaluated and revised where appropriate.

There have been no previous similar events.

END OF ABSTRACT

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I. DESCRIPTION OF WHAT OCCURRED:

A. Initial Conditions:

On November 16, 1988, Palo Verde Unit 2 was in Mode 1 (POWER OPERATION) performing a plant shutdown to investigate and repair a Reactor Coolant System (RCS)(AB) leak. Immediately prior to the reactor (RCT)(AC) trip discussed below, reactor power was approximately 10 percent.

B. Reportable Event Description (Including Dates and Approximate Times of Major Occurrences):

Event Classification: Automatic actuation of the Reactor Protection System (RPS)(JC).

On November 16, 1988 at approximately 0237 MST Palo Verde Unit 2 was in Mode 1 (POWER OPERATION) at approximately 10 percent power when a reactor trip occurred due to low steam generator (SG)(AB) water level. The low steam generator water level resulted from main feedwater pump (P)(SJ) speed being inadequate to supply feedwater to the steam generators during a power reduction. There were no engineered safety features (ESF)(JE) actuations and none were necessary. The event was properly diagnosed as an Uncomplicated Reactor Trip. At approximately 0247 MST both steam generator levels had been raised to above their trip setpoints and stable conditions were achieved. The event lasted approximately 10 minutes.

Prior to the reactor trip, Palo Verde Unit 2 was being shut down to investigate and repair the cause of an unidentified leak. Initial indications of a leak became apparent on November 7, 1988 when day-shift operations personnel (utility, licensed and non-licensed) noted an unexplained increase in the reactor cavity sump (WK) level over the previous few days. Initial estimates of the leakage into the sump were approximately 0.5 gallon per hour (0.008 gallon per minute).

It should be noted that Technical Specification 3.4.5.2 allows 1.0 gallon per minute (gpm) unidentified Reactor Coolant System (RCS) leakage. Therefore, assuming that the leakage into the reactor cavity sump was unidentified reactor coolant system leakage, continued plant operation was allowed.

Investigation was initiated to identify the cause of the leakage. On November 8, 1988, troubleshooting was performed on the reactor cavity sump level indicator (LI)(WK) and it was determined that the level indicator was operating properly. Furthermore, no significant trends or changes were noted in containment (NH)

atmosphere samples for iodine, noble gas, or particulate radioactivity. Therefore, preparations were made to enter the containment to visually search for the cause of the leakage.

The initial containment entry to investigate the cause of the leakage was made on November 9, 1988. Initial attempts were unsuccessful in identifying the source of the leakage so the investigation continued. On November 15, 1988, a steam leak was discovered in the vicinity of the number 1 RCS hot leg. The exact source of the leakage was indeterminate due to the existing radiological conditions and presence of water vapor. Therefore, management prudently decided to shut down Unit 2 to facilitate further inspections and repairs.

At approximately 2000 MST on November 15, 1988, a power reduction from 100 percent power was commenced. At approximately 15 percent power, the Feedwater Control System (FW)(JB) automatically redirected feedwater flow from the steam generator's economizer region to the downcomer region (i.e., both steam generator economizer regulating valves (FCV)(SJ) shut, both downcomer regulating valves (FCV)(SJ) opened). Concurrent with the redirection of feedwater flow, the operating main feedwater pump speed reduced to approximately 3759 revolutions per minute (RPM), and the Feedwater Control System control methodology changed such that steam generator level controlled the amount of feedwater flow (vice a combination of measured feedwater and steam flow as well as the steam generator level). As a result of the main feedwater pump speed decreasing, inadequate discharge head was developed to overcome steam generator pressure at the low power conditions. Steam generator levels decreased until a reactor trip occurred at approximately 0237 MST on November 16, 1988. The reactor trip resulted from a steam generator number 2 low level trip signal. The event was properly diagnosed as an uncomplicated reactor trip.

Following the trip, the "B" Train Essential Auxiliary Feedwater Pump (BA)(P) and the Non-essential Auxiliary Feedwater Pump were manually started to feed both steam generators. At approximately 0247 MST both steam generator levels had been raised to above their trip setpoints and the event was terminated.

On November 16, 1988, the RCS leak was determined to be valve packing leakage from one of the Plant Protection System (JS) Channel D Steam Generator differential pressure transmitter (PDT) root isolation valves (ISV). The valve was appropriately repaired on November 19, 1988.

During the inspection being conducted to identify the source of the RCS leakage, it was discovered that the water from the leaking valve was flowing through small cracks in a concrete wall for the incore instrumentation (IG) chase. An engineering evaluation of this condition was initiated.

C. Status of structures, systems, or components that were inoperable at the start of the event that contributed to the event:

Not applicable - no components, systems, or structures were inoperable at the start that contributed to the event.

D. Cause of each component or system failure, if known:

The failed bolt was manufactured from carbon steel and was exposed to boric acid from the reactor coolant system. The boric acid degraded the bolt until tensile forces resulted in failure.

E. Failure mode, mechanism, and effect of each failed component, if known:

The RCS leak resulted from a broken packing gland follower bolt. The broken bolt allowed the packing gland follower to cant which reduced the compression on the packing and resulted in excess packing leakage.

F. For failures of components with multiple functions, list of systems or secondary functions, that were also affected:

Not applicable - there were no component failures with multiple functions.

G. For failure that rendered a train of a safety system inoperable, estimated time elapsed from the discovery of the failure until the train was returned to service:

Not applicable - no safety systems were rendered inoperable. The failed bolting resulted in packing leakage significantly below Technical Specification 3.4.5.2 limits.

H. Method of discovery of each component or system failure or procedural error:

The broken bolting was discovered during ANPP's investigation into the reason for the reactor cavity sump level increase (see Section I.B). There were no system failures. Procedural inadequacies were discovered as a result of ANPP's Post Trip Review process.

I. Cause of Event:

There were
oncurrent contributory causes which resulted in the reactor trip.
The first cause is that the automatic Feedwater Control System response was not adequate in maintaining a sufficient feedwater pump speed following the period that feedwater flow to the steam generators shifted from the economizer region to the downcomer region. In part, this was due to main feedwater pump speed adjustments which were made by operations personnel during normal power operations (The adjustments procedurally were required to minimize economizer valve oscillations). The adjustments, coupled with a programmed main feedwater pump speed limitation when feedwater flow shifts to the downcomer region, resulted in pump speed being too slow for the existing plant conditions. It should be noted that a site modification was installed in July 1988 which lowered the main feedwater pump minimum speed. The site modification was originally implemented to resolve Feedwater Control System (FWCS) performance problems in Unit 1. It had been necessary to take manual control of the FWCS, vice leaving it in automatic, during low power operations in order to prevent overfeeding the steam generators. The site modification was also prepared and implemented in Units 2 and 3 and the resulting FWCS was more versatile; however, the system potentially required operator action during certain plant conditions.

Inadequate reviews were conducted for identifying and delineating the necessary operator actions. This was especially important for Unit 2 since the steam pressure is approximately 20 psi higher in Unit 2 than Unit 1 during low power operations.

Another contributory cause is that procedural controls utilized at low power operations (i.e., below twenty percent power) did not contain explicit guidance for ensuring that the proper adjustments were made to the automatic main feedwater pump speed control to compensate for the adjustments made at normal power operations. As a result of implementing the site modification to reduce minimum feedwater pump speed, procedure revisions were not initiated which would have provided additional guidance for ensuring adequate feedwater supply.

Another concern, which may have had an impact on this event, involves operator performance. Control Room operating personnel (utility, licensed) on-shift during the power reduction did not take the appropriate compensatory measures which would have maintained main feedwater pump speed at an adequate level for feeding the steam generators. This would have required adding significant positive bias to the feed pump speed controller which was not addressed in the procedure. If additional information regarding this concern is identified which would significantly alter the perception of this event, a supplement to this report will be submitted.

There were no unusual characteristics of the work location which contributed to this event. Except as noted above, procedural controls have been determined to be adequate.

The concrete walls surrounding the incore instrumentation chase consist of mass concrete placed to form the chase, access shaft, ventilation shaft, and reactor cavity. The walls are variable thickness and are load bearing in that they transfer loads from the primary shield above to the containment basemat. The walls are under constant compression with relatively low stress levels. There are no flexual or tensile stress loads. The cracks identified in the incore chase concrete wall are vertical cracks probably induced by mass volume changes which resulted from temperature changes. This type of crack formation is not unusual.

J. Safety System Response:

The following manual and automatic safety system responses occurred:

Plant Protection System automatic initiation of reactor trip.

Essential Auxiliary Feedwater Pump "B" was manually started by Control Room personnel.

K. Failed Component Information:

The broken packing gland follower bolt was supplied as part of the valve manufactured by Borg Warner. The model number of the valve is 77540. The failed bolt is manufactured from A540 Grade B23 carbon steel material.

II. ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATIONS OF THIS EVENT:

There were no safety consequences or implications resulting from this event. There was no impact on public health and safety. The uncomplicated reactor trip occurred per design as a result of the low steam generator water level. Water level remained above the point which would have required an automatic Auxiliary Feedwater Actuation (JE)(BA). Adequate heat removal capabilities existed throughout the event. This event could not have occurred at higher power levels because the Feedwater Control System swapover from the economizer to the downcomer is controlled by Nuclear Instrumentation at 15 percent power.

There were no safety consequences or implications resulting from the bolting failure on the differential pressure transmitter isolation valve. Leakage through the valve's packing as a result of the failed bolt remained below Technical Specification limits for continuous operation. There are two packing gland follower bolts on the affected valve. The other bolt remained functional throughout the event.

There are no safety consequences or implications resulting from the cracks in the incore instrument chase concrete wall. The type of crack formation is not unusual. Since the cracks are vertical, no load transfer path is interrupted and no other design function is compromised. The structural design basis of the containment internal structure is unchanged by the cracks.

III. CORRECTIVE ACTIONS:

A. Immediate:

The failed bolt as well as the other remaining bolt on the instrument root valve have been replaced.

B. Action to Prevent Recurrence:

As action to prevent recurrence, additional instructions have been included in operating procedures to ensure that operations personnel take the appropriate measures for maintaining adequate main feedwater pump speed during low power operations.

An engineering evaluation of Unit 2's automatic feedwater control system operation will be performed. As an initial result of the Engineering Evaluation, an enhancement to the preventive maintenance task for the feedpump governor is being implemented.

A Human Performance Evaluation is being performed to address factors which contributed to the operations personnel performance concern. Additionally, this event will be reviewed by the appropriate operations department personnel from Units 1, 2, and 3 during normally scheduled periodic training. If additional corrective action is identified as a result of the Human Performance Evaluation which significantly alter impact the perception of this event, a supplement to this report will be issued.

As discussed in Section I.I, the appropriate procedure changes did not get implemented as a result of the site modification. As a corrective action, the current site modification administrative controls will be evaluated and improved where appropriate. The system engineer program and the system engineer/Plant Standards and Control interface will be evaluated to determine if improvements can be implemented to ensure that necessary procedure changes are incorporated following plant modifications. A representative sample of current site modifications will be reviewed to determine if additional procedure revisions are necessary.

Concerning the bolt failures discussed in Section I.B, the problem with boric acid causing premature failures had been previously identified. An engineering evaluation had been performed and as corrective action, a new bolting material was specified (ASTM A564 TP 630). The bolting is being changed out on an "as-needed" basis.

Concerning the cracks in the incore instrument chase concrete wall, a procedure for sealing the cracks is being developed. The cracks will be sealed during Unit 2's next refueling outage. It should be noted that this is a long-term solution for corrosion protection. A design or structural repair is not required.

IV. PREVIOUS SIMILAR EVENTS:

There have been no previous similar events reported pursuant to 10CFR50.73. It should be noted that other reactor trips have been reported which resulted from feedwater flow problems; however, none involve the sequence of events or root cause described in this LER.

Arizona Nuclear Power Project
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192-00437-JGH/TDS/DAJ
December 14, 1988

U. S. Nuclear Regulatory Commission
NRC Document Control Desk
Washington, D.C. 20555

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 2
Docket No. STN 50-529 (License No. NPF-51)
Licensee Event Report 88-014-00
File: 88-020-404

Attached please find Licensee Event Report (LER) No. 88-014-00 prepared and submitted pursuant to 10CFR 50.73. In accordance with 10CFR 50.73(d), we are herewith forwarding a copy of the LER to the Regional Administrator of the Region V office.

If you have any questions, please contact T. D. Shriver, Compliance Manager at (602) 393-2521.

Very truly yours,

J. G. Vice President
Nuclear Production
JGH/TDS/DAJ/kj

Attachment

cc: D. B. Karner (all w/a)
E. E. Van Brunt, Jr.
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INPO Records Center

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